Enclosure 1

Containment Failure Matrix and Radiological Source Term for the Millstone-3 DES

Outline

4/3/84

I. Introduction

The Draft Environmental Statement (DES) for the Millstone Unit 3 will include a severe accident risk estimate based on site consequence analyses performed by the Accident Evaluation Branch (AEB). As input to these calculations, the Reactor Systems Branch (RSB) and the Containment Systems Branch (CSB) are providing AEB with an estimate of the conditional probabilities of various potential containment building failure modes (C-matrix). The radiological source term is being specified jointly by RSB and AEB. The Reliability and Risk Assessment Branch (RRAB) will eventually provide the plant damage state probabilities based on a review of Millstone-3 Probabilistic Safety Study (MPSS)[1] by RRAB staff and contractors at Lawrence Livermore National Laboratory (LLNL).[2] The data presented herein is preliminary and may be changed prior to final input to the DES and refers only to internally initiated events.

The data presented in this enclosure are based largely on the MPSS, which has been reviewed by RSB and CSB staff and contractors at Brookhaven National Laboratory (Reference 3). Several adjustments to the MPSS results have been made, for reasons which will be described in the following sections.

II. Description of Plant Damage States

In the Millstone Probabilistic Safety Study (MPSS), each core melt accident sequence is assigned to one of the twenty-seven plant damage states described in Table 1. Summation over all of the frequencies of core melt accidents associated with a given plant damage state yields the annual frequency of the damage state. These frequencies, which are listed in Table 2, are preliminary and based on the LLNL[2] review. They are currently under review by RRAB staff and may be adjusted prior to input to the DES. The original MPSS plant damage state frequencies are also included in Table 2 for reference. Note that in the MPSS, twenty-seven plant damage state frequencies were identified, whereas in the LLNL review, only seventeen plant damage state frequencies were given. The LLNL review eliminated twelve damage states (namely, AEC', AE, ALC", AL, SE, S'E, SLC", SL, V2E, V2LC', V2LC" and V2L) from further consideration because of low probability ($<10^{-7}$) but also added two additional damage states (namely, S'EC and TLC).

The plant damage states classify events according to three parameters;

(1) Initiating Event, namely:

A, large break Loss-Of-Coolant Accidents (LOCAs)

S, small break LOCAs

S', incore instrument tube LOCA

T, transients

V2, Steam Generator Tube Rupture (SGTR)

V, Interfacing Systems LOCA

(2) Timing of Core Melt, namely:

E, failure of Emergency Core Cooling Injection (ECCI)

L, failure of ECC recirculation

(3) Status of Containment Heat Removal (CHR)

__, complete loss of Containment Sprays (CS)

C', loss of recirculation CS

C", loss of quench CS

C, all spray systems available

In the following sections the process of relating the plant states to potential containment building failure modes and fission product release characteristics is described.

III. Containment Failure Probabilities (C-Matrix)

In the MPSS, the twenty-seven plant states identified in Table 1 were related to potential containment building failure modes by using containment event trees. It was considered unnecessary to analyze each individual plant state because of common characteristics relative to primary system response, containment response, and source term. The primary system response characteristics were grouped using accident sequence classes (A-G in the MPSS). Accident sequences were classified in the MPSS according to:

(1) the initiating event,

(2) time of onset of fuel melt, and

(3) RCS conditions at time of vessel failure, particularly RCS pressure.

Five of the sequence classes (A-E) required further analysis to characterize the containment response. Accident classes F (interfacing system LOCA) and G (ruptured steam generator tube) bypass the containment and hence were allocated directly to an appropriate release path and fission product source term.

Characterization of containment response for the five accident classes (A-E) required four possible combinations of quench spray system and recirculation spray system operation. These quench and recirculation spray system combinations are:

(1) both quench sprays and recirculation sprays on

(2) both sprays off

(3) quench sprays on, recirculation sprays off

(4) recirculation sprays on, quench sprays off

This characterization by accident sequence and containment response for five of the accident classes defines twenty distinct accident groups or categories. Again, because of common characteristics, it was not considered necessary to assess all of the possibilities and hence only ten containment response classes were quantified using containment event trees in the MPSS. These containment response classes are defined in Table 3.

Table 4 summarizes the containment response classes with the corresponding plant damage states and their associated mean frequencies as provided in the LLNL review (see Table 2).

Therefore, these containment response classes can be related to the radiological release categories to form the containment matrix.

The quantification of the MPSS containment event trees was a significant task, and it was necessary to use a computer code, ARBRE, to group the various path probabilities into the thirteen release categories.[1] However, the containment matrix 'C' is a concise summary of the quantification process.

Table 5 is a reproduction of the 'C' matrix for the MPSS.[1] It lists the conditional probabilities of the release categories (defined in Table 6) given the plant damage state, with the plant damage states defined earlier in Table 2.

A simplification to the C-matrix is obtained in Table 7 by disregarding all of the very low probability values ($CP<10^{-2}$). This simplification is not expected to influence the risk calculations.

Table 7 indicates that the containment classes 1 through 3 lead to intermediate and late overpressure failures or basement melt-through in the absence of CHR operation, with an early failure being more likely as a result of hydrogen burn for classes 1 and 3. Furthermore, the containment response classes 4, 5, 7, 8, and 10 are dominated by intermediate or late overpressure failures without full CHR operation, with basemat penetration being less likely. However, successful operation of containment recirculation spray system leads to basemat failure for class 9 states. It should also be noted that the most probable sequence (class 6) leads to the lowest failure probability.

IV. Source Term Probabilities

In Table 7, conditional probabilities for the various release categories given a containment response class were assigned. In order to determine the frequency of occurrence of the source terms summarized in Table 8, the containment class frequencies listed in Table 4 must be multiplied by the conditional probability of the containment failure modes given in Table 7. These frequencies are included in the source term characterization in Section V.

V. Radiological Source Term

V.1 MPSS Release Fractions

For most of the release categories, the applicant's evaluation of radionuclide release fractions was based on CORRAL-II calculations. For a few release categories, the release fractions were taken directly from WASH-1400. These two approaches are consistent insofar as both account for the same mechanisms of fission product release, transport, and deposition. Three components of release from the core were included: gap release, core melt release, and vaporization release. Radionuclide attenuation due to deposition on containment surfaces, gravitational settling and washout by containment sprays was calculated.

Because of uncertainties in the chemical form of iodine, two sets of release fractions were calculated; one characteristic of gaseous elemental iodine and one representative of CsI aerosol. The latter source term was used for all calculations in the MPSS. The principal difference between the two options is that the aerosol model yields significantly higher iodine releases for release categories M-5, M-6, and M-7; the intermediate and late overpressurization failure modes without sprays. Because M-7 is the most likely mode of failure, these differences could be important.

Since the publication of WASH-1400, it has become apparent that iodine will have a strong tendency to form Cesium Iodide and subsequently adhere to aerosols. However, the Accident Source Term Program Office (ASTPO) is currently in the process of assessing this question, as well as numerous other issues related to the source term. Until these results can be quantified and submitted to peer review, the agency will continue to base licensing decisions on WASH-1400 methodology. Consequently, we have used the release fractions characteristic of elemental iodine (Table 9) as the starting point for our review.

Comparisons of Table 9 with other studies performed with WASH-1400 methodology have led us to conclude that the iodine releases in M-5, M-6, and M-7 are too low. In references [4] and [5], the iodine releases for late overpressure failure at Indian Point were an order of magnitude higher than the MPSS results (Table 10). The releases of all other radionuclides were of comparable magnitude. We have used the higher iodine release fractions for this DES input (refer to Section V.3).

V.2 Discrete Probability Distributions (DPD) Used in the MPSS

The release fractions in Table 9 do not reflect all mechanisms of source term attenuation. Retention of fission products in the primary system was not credited. Furthermore, the enhancement of gravitational settling in containment due to aerosol agglomeration was not included. To account for these factors and their associated uncertainties, the applicant employed the method of discrete probability distributions. In this method, the actual release fractions for a given release category can assume values which are a fraction (F) of the values given in Table 9. The allowed fractions are 1, 1/2, 1/4, 1/10 and 1/100. A probability (P) is associated with each F, and the probabilities are different for each release category (Table 11). For example, in a failure to isolate containment (M-4), there is an assumed 40% probability that F is equal to unity, and a 60% probability that F is 1/2. This small reduction in fission product release reflects an assumed retention of fission products in the primary system, but very little effect of agglomeration. For late failure without sprays (M-7), agglomeration is assumed to play a significant role, and the source term is reduced by a factor of 1/10 to 1/100. The values of F and P are based largely on engineering judgment. In all cases, the discrete probability distributions lead to a reduction in the radiological source term.

We have examined the DPD methodology and concluded that it should not be factored into the release fractions used for the DES. Fission product retention in the primary system and aerosol agglomeration in containment are credible mechanisms for fission product attenuation, and are currently under study by the Accident Source Term Program Office (ASTPO). Until the ASTPO evaluation of the existence and magnitude of these mechanisms is complete, we will not have a sound basis for quantifying the reduction in the source term. We recognize that the decision not to factor in the DPD's represents a conservative approach to the source term.

V.3 Suggested Source Terms for Input to Millstone-3 DES

In this section, the approach utilized to determine the fraction of fission products originally in the core and leaked to the outside environment will be outlined. The fission product source to the environment, as calculated by this approach, will be compared with those for similar plants. The calculations to be included in this comparison are those done for the Zion and Indian Point Probabilistic Risk Assessments, (ZPSS[6] and IPPSS,[5] respectively), and the Indian Point Study (IPS) carried out for the NRC and presented as testimony[4] at the Indian Point hearings. These calculations are based on the methods used in the Reactor Safety Study (RSS), which was published as WASH-1400.[7]

In the RSS, the CORRAL-II code was the mathematical model used to determine fission product leakage to the environment. This code takes input from the thermal-hydraulic analysis carried out for the containment atmosphere. In addition, it needs the time dependent emission of fission products. The fission product release is divided up into three phases, namely, Gap, Melt, and Vaporization releases. The time dependence of these phases is determined by the core heatup, primary system failure and core/concrete interactions. In all, thirteen releases were determined in the MPSS using these methods ranging from the containment bypass sequence (V-sequence) to the no fail sequence. The results are shown on Table 9.

Some of the thirteen MPSS releases outlined in Table 9, namely M-1A (PWR-2), M-10 (PWR-6), and M-11 (PWR-7) are identical in both fractional release and timing to equivalent PWR releases in the RSS. The release M-1B, which corresponds to a steam generator tube rupture, is determined by dividing PWR-2 or M-1A by ten. Noble gases and organic iodine are not subject to this reduction in release.

There are two areas of significant disagreement between the MPSS and the staff review. These are the iodine release for the overpressurization failure sequences (M-5, M-7) and the energy of release for these sequences. It is felt that the fraction of iodine released to the environment should be increased from .015 to .1 for these sequences. This recommendation is based on a comparison between the MPSS results and those determined in the IPPSS and IPS. Shown in Table 10 are the fractions of fission products released for the M-5 and M-7 sequences compared with similar sequences in the IPPSS and IPS. From an inspection of Table 10, it can be seen that the release fractions for all the species agree well, except for iodine.

The energy of release for the overpressurization failures are high compared to those used in the RSS, IPPSS, and IPS. In fact, the values are more characteristic of the values used for a steam explosion failure mode in the RSS. The effect of a high energy of release on the plume is to lift it higher into the atmosphere and thus spread it over a larger area. By comparing the MPSS values with those used in the above studies, it is felt that the energy of release should be reduced to 150×10^8 Btu/hr. This value is higher than the values used in the IPPSS and IPS, however, it is felt to be a reasonable value for the overpressurization failure mode.

In Tables 12-15, release characterizations for the dominant sequences are shown.

Table 12 shows release fractions and timing for two containment bypass sequences; the first being an interfacing loss of coolant accident (Event V) and the second representing a steam generator tube failure (Event V2).

Shown on Table 13 are release fractions for overpressure failures of the containment during various time frames ranging from 4.3 hrs to 20.1 hrs. No spray operation is assumed during these sequences.

Tables 14 and 15 show release fractions and timing for basemat penetration and no containment failure, respectively. Table 16 shows the release fraction to be used for a steam explosion initiated failure mode. This release fraction and timing are based on the WASH-1400 PWR-1 release. The frequency of a steam explosion release was assumed to be 10^{-4} of the total core melt frequency, which is consistent with previous DES analyses (e.g., Limerick).

VI. Further Work

The source terms given in Tables 12 through 15 represent our recommended input to the DES for Millstone-3 at this time. The source terms are based, in large part, on the MPSS and on a rather limited review of the MPSS by the NRC staff and contractors. However, the Millstone-3 DES has been postponed and thus provides additional time to refine our source term estimates. In this section, we indicate those areas in which our Millstone-3 source term estimates will receive further investigation. The results of these investigations will be factored into our final report.

External Events - The present assessment is limited to the internal initiating events; however, the containment response to accidents initiated by external initiating events (fires, floods, and seismic events) must also be reviewed.

Hydrogen - In the MPSS for accidental sequences without CHR, the conditional probability of an intermediate failure (M6) from a H_2 burn relative to a late failure (M7) due to overpressurization, varies significantly depending on the initiator (LOCA vs. Transient). If the accident sequence is initiated by a large break LOCA, then the conditional probability of a H_2 burn failure mode is 0.62 compared with 0.06 for a small break LOCA, and negligible probability for sequences initiated by transients. In the IPPSS, ZPSS and IPS, no such distinction was made for these accident sequences. We therefore will determine if we can support the conditional probabilities of a H_2-burn failure for containment classes 1-4.

Containment Failure Distribution - In MPSS-3, the containment failure probability distribution has been calculated. This failure distribution will be carefully evaluated.

Debris Quenching - The quantitative significance of debris quenching in the reactor cavity will be examined.

Elemental Iodine - The acceptability of the relatively low release fraction of elemental iodine for sequences M-5, M-7, and M-9 compared to releases for similar sequences determined by other investigators (IPPSS) will be determined.

Energy of Release - The higher energy of release for the overpressurization failures, compared to energy releases for similar failure modes determined by other analysts (IPPSS) will be examined.

Warning Time - For sequence M-6, the release time is 4.3 hrs and the warning time is 4.1 hrs. This timing implies that the operating staff responds quite rapidly to the accident. The feasibility of such a rapid response and its acceptability for use in the MPSS-3 will be investigated.

The LLNL review also introduced two new plant states (namely, S'EC and TLC), which we binned into containment class 6. We will confirm that this is an appropriate containment class for these sequences. In addition, the difference in response for TEC' and SEC' will be resolved.

VII. References

(1) "Millstone Unit 3 Probabilistic Safety Study," Northeast Utilities, August 1983.

(2) "A Review of the Millstone-3 Probabilistic Safety Study," Incomplete Preliminary Draft, January 25, 1984.

(3) M. Khatib-Rahbar, H. Ludewig, and W. T. Pratt, "Preliminary Review and Evaluation of the Millstone-3 Probabilistic Safety Study," Brookhaven National Laboratory, Informal Report, December 1983.

(4) Direct Testimony of J. F. Meyer and W. T. Pratt concerning Commission Question 1, Indian Point Hearings, Docket Numbers 50-247 and 50-286, 1983.

(5) "Indian Point Probabilistic Safety Study," Power Authority of the State of New York and Consolidated Edison Co., March 1982.

(6) "Zion Probabilistic Safety Study," Commonwealth Edison Company, September 1981.

(7) Reactor Safety Study, "An Assessment of Accident Risks in U.S. Commercial Nuclear Power Plants", WASH-1400, NUREG/75-014, October 1975.

Table 1 Notation and definitions for plant states (internal)

Symbol	Description
AEC	Large LOCA, Early Melt
AEC'	Large LOCA, Early Melt, Failure of Recirculation Spray
AE	Large LOCA, Early Melt, No Containment Cooling
ALC	Large LOCA, Late Melt
ALC'	Large LOCA, Late Melt, Failure of Recirculation Spray
ALC"	Large LOCA, Late Melt, Failure of Quench Spray
AL	Large LOCA, Late Melt, No Containment Cooling
SEC	Small LOCA, Early Melt
SEC'	Small LOCA, Early Melt, Failure of Recirculation Spray
SE	Small LOCA, Early Melt, No Containment Cooling
S'E	Incore Instrument Tube LOCA, Early Melt, No Containment Cooling
SLC	Small LOCA, Late Melt
SLC'	Small LOCA, Late Melt, Failure of Recirculation Spray
SLC"	Small LOCA, Late Melt, Failure of Quench Spray
SL	Small LOCA, Late Melt, No Containment Cooling
S'L	Incore Instrument Tube LOCA, Late Melt, No Containment Cooling
TEC	Transient, Early Melt
TEC'	Transient, Early Melt, Failure of Recirculation Spray
TE	Transient, Early Melt, No Containment Cooling
V2EC	Steam Generator Tube Rupture, Steam Leak, Early Melt
V2EC'	SGTR, Steam Leak, Early Melt, Failure of Recirculation Spray
V2E	SGTR, Steam Leak, Early Melt, No Containment Cooling
V2LC	SGTR, Steam Leak, Late Melt
V2LC'	SGTR, Steam Leak, Late Melt, Failure of Recirculation Spray
V2LC"	SGTR, Steam Leak, Late Melt, Failure of Quench Spray
V2L	SGTR, Steam Leak, Late Melt, No Containment Cooling
V	Interfacing Systems LOCA

Table 2 Plant damage state frequencies for internal events (per reactor-year)

Symbol	MPSS (Mean)	Provided by RRAB
AEC	1.92E-06	8E-7
AEC'	4.17E-09	*
AE	2.68E-09	*
ALC	5.44E-06	2E-6
ALC'	4.88E-7	1E-7
ALC"	3.42E-09	*
AL	3.36E-10	*
SEC	1.12E-06	2E-5
SEC'	2.76E-09	6E-7
SE	1.17E-07	*
S'EC	-	4E-7
S'E	1.83E-09	*
SLC	9.81E-06	1E-4
SLC'	4.79E-07	1E-5
SLC"	5.77E-08	*
SL	2.73E-09	*
S'L	3.35E-10	1E-7
TEC	1.81E-05	4E-5
TEC'	3.46E-07	2E-7
TE	5.31E-06	7E-6
TLC	-	4E-5
V2EC	1.11E-07	4E-6
V2EC'	1.03E-09	3E-7
V2E	1.29E-08	*
V2LC	2.76E-09	2E-7
V2LC'	1.49E-10	*
V2LC"	1.77E-11	*
V2L	8.40E-13	*
V	1.90E-06	4E-7
TOTAL	4.53E-05	2.3E-4

*Indicates frequency values <10^{-7}.

Table 3 Containment response classes

Class	Dominant Sequence	Reference Definitions
1	AE	Initiating event is typically a large break LOCA without safety injection and without minimum containment safeguards operating throughout the transient.
2	SE	Same as the AE sequence except that the initiating event is typically a small break LOCA or transient event. Note that the containment sprays do not operate.
3	AL	Same as the AE sequence except that safety injection is initiated but operate only until switchover to recirculation is attempted, at which time it becomes inoperative for the remainder of the transient.
4	TE	The initiating event is typically a transient in which all power is lost. There would therefore be no safety injection and no containment safeguards initiation at any time during the transient.
5	SL	Same as the AL sequence except that the initiating event is typically a small break LOCA or transient event. Note that the containment sprays are actuated but do not deliver water to the spray headers.
6	TEC	Same as the TE sequence except that all containment heat removal systems are available.
7	TEC'	Same as TE sequence (Class 4) except that AC power is available and containment quench spray system is functioning.
8	SEC'	Same as SE sequence (Class 2) except that containment quench spray system is functioning.
9	TEC"	Same as TE sequence (Class 4) except that AC power is available and recirculation spray system is functional.
10	S'L	Same as SL sequence (Class 5) except that rupture is as incore instrumentation tube rupture.

Table 4 Containment class mean frequencies for internal events (per reactor year)

Containment Class	Plant Damage States	Mean Frequency (yr^{-1})
1	AE	*
2	SE	*
3	AL	*
4	TE	7.0E-6
5	SL	*
6	AEC, ALC, SEC, SLC, SEC, TEC, TLC, S'EC	2.03E-4
7	TEC', SLC'	1.02E-5
8	AEC', ALC', SEC'	7.0E-7
9	AEC", ALC", SEC", SLC", TEC"	*
10	S'E, S'L	1.0E-7
	V2EC, V2EC', V2E, V2LC, V2LC', V2LC", V2L	4.5E-6
	V	4.0E-7

*Indicates frequency value less than 10^{-7}.

Table 5 Reproduced from MPSS Table 4.7.2-2

Table 6 Notation and definitions for release categories

Release Category	Description
M1A	Containment Bypass, V-Sequence
M1B	Containment Bypass, SGTR
M2	Early Failure/Early Melt, No Sprays
M3	Early Failure/Late Melt, No Sprays
M4	Containment Isolation Failure
M5	Intermediate Failure/Late Melt, No Sprays
M6	Intermediate Failure/Early Melt, No Sprays
M7	Late Failure, No Sprays
M8	Intermediate Failure With Sprays
M9	Late Failure With Sprays
M10	Basemat Failure, No Sprays
M11	Basemat Failure With Sprays
M12	No Containment Failure

Table 7 Simplified containment matrix for MPSS

Containment Response Class	M1A	M1B	M5	M6	M7	M10	M11	M12
1				0.62	0.29	0.09		
2				0.06	0.89	0.05		
3				0.54	0.35	0.11		
4					0.90	0.10		
5			0.01		0.79	0.20		
6							0.05	0.95
7					1.0			
8					1.0			
9							0.99	0.01
10					0.99	0.01		
V	1.0							
V2		1.0						

Table 8 Source term frequencies

Containment Response Class	M1A	M1B	M5	M6	M7	M10	M11	M12	Frequency (yr^{-1})
1				*	*	*			*
2				*	*	*			*
3				*	*	*			*
4					6.3E-6	7.0E-7			7.0E-6
5			*		*	*			*
6							1.01E-5	1.93E-4	2.04E-4
7					1.02E-5				1.02E-5
8					7.0E-7				7.0E-7
9						*	*	*	*
10				*	*				*
V	4.0E-7								4.0E-7
V2		4.5E-6							4.5E-6
Release (yr^{-1}) Frequency	4.0E-7	4.5E-6	*	*	1.72E-5	7.0E-7	1.01E-5	1.93E-4	

*Indicates frequency value less than 10^{-7}

Table 9 - Reproduced from MPSS

Table 10 Intermediate and late overpressurization (no sprays)

	Sequence			
	MPSS M-5	MPSS M-7	IPS[4] TMLB' - δ	IPPSS[5] 2RW*
Xe-Kr	.9	.9	.96	1.0
OI+I	.016	.015	1.05(-1)	9.3(-2)
Cs-Rb	.5	.3	.34	.26
Te-Sb	.5	.3	.38	.44
Ba-Sr	5(-2)	3(-2)	3.7(-2)	2.5(-2)
Ru	4(-2)	2(-2)	2.9(-2)	2.9(-2)
La	6(-3)	4(-3)	4.9(-3)	1.0(-2)

Table 11 MPSS release category DPDs

Release Category	Discrete Probability Distributions				
	F* 1	1/2	1/4	1/10	1/100
M-1A	0.17**	0.55	0.28	0	0
M-2	0.25	0	0.25	0.50	0
M-3	0.0	0	0.06	0.63	0.31
M-4	0.40	0.60	0	0	0
M-5	0.0	0.0	0.05	0.64	0.31
M-6	0.11	0.14	0.27	0.48	0
M-7	0	0	0	0.11	0.89

*Release Fraction (F)

**Probability Values (P)

Table 12 Containment bypass sequences

Failure Mode and Release Paths	M-1A	M-1B
Xe-Kr	9(-1)	9(-1)
I+OI	7.07(-1)	7.07(-1)
Cs-Rb	5(-1)	5(-2)
Te-Sb	3(-1)	3(-2)
Ba-Sr	6(-2)	6(-3)
Ru	2(-2)	2(-3)
La	4(-3)	4(-4)
Release Time (hr)	2.5	2.5
Warning Time (hr)	1.0	1.0
Duration (hr)	1.0	1.0
Energy (10^6 Btu/hr)	20.0	20.0
Probability	4E-7	4.5E-6

Table 13 Intermediate and late overpressurization failure

	M-5	M-6	M-7
Xe-Kr	9(-1)	9(-1)	9(-1)
OI+I	0.1	0.1	0.1
Cs-Rb	5(-1)	5(-1)	3(-1)
Te-Sb	5(-1)	5(-1)	3(-1)
Ba-Sr	5(-2)	5(-2)	3(-2)
Ru	4(-2)	4(-2)	2(-2)
La	6(-3)	7(-3)	4(-3)
Release Time (hr)	8.3	4.3	20.1
Warning Time (hr)	4.1	4.1	16.0
Duration (hr)	0.5	0.5	0.5
Energy (10^6 Btu/hr)	150	150	150
Probability	$<10^{-7}$	$<10^{-7}$	1.72×10^{-5}

Table 14 Basemat penetration

Failure Mode and Release Paths	M-10	M-11
Xe-Kr	3(-1)	6(-3)
I+OI	2.8(-3)	4(-5)
Cs-Rb	8(-4)	1(-5)
Te-Sb	1(-3)	2(-5)
Ba-Sr	9(-5)	1(-6)
Ru	7(-5)	1(-6)
La	1(-5)	2(-7)
Release Time (hr)	95	95
Warning Time (hr)	80	80
Duration (hr)	10	10
Energy (10^6 Btu/hr)	-	-
Probability	7×10^{-7}	1.01×10^{-5}

Table 15 No containment failure

Failure Mode and Release Path	M-12
Xe-Kr	1(-3)
OI+I	1.5(-5)
Cs-Rb	1(-6)
Te-Sb	9(-7)
Ba-Sr	2(-7)
Ru	8(-8)
La	1(-8)
Release Time (hr)	.5
Warning Time (hr)	-
Duration (hr)	5.0
Energy (10^6 Btu/hr)	-
Probability	1.93×10^{-4}

Table 16 Steam explosion failure mode

Failure Mode and Release Path	PWR 1
Xe-Kr	9(-1)
OI+I	7(-1)
Cs-Rb	4(-1)
Te-Sb	4(-1)
Ba-Sr	5(-2)
Ru	4(-1)
La	3(-3)
Release Time (hr)	2.5
Warning Time (hr)	1.0
Duration (hr)	0.5
Energy (10^6 Btu/hr)	520
Probability	2×10^{-8}

ENCLOSURE 2

REQUEST FOR ADDITIONAL INFORMATION

MILLSTONE NUCLEAR POWER STATION, UNIT 3

NORTHEAST NUCLEAR ENERGY COMPANY

DOCKET NO. 50-423

REQUEST FOR ADDITIONAL INFORMATION
MILLSTONE NUCLEAR POWER STATION, UNIT 3
DOCKET NO. 50-423

720.0 Reliability and Risk Assessment Branch

720.92 The staff needs further information to establish the validity of your response to question 720.77. You have stated that further analysis by Dames and Moore has indicated that events in the magnitude range of 5.3 to 6.3 dominate the hazard even for ground motions as large as 0.6g and higher.

(1) How can this be the case considering the peak ground acceleration truncation which you assume? Provide the specific Dames and Moore analysis which you have used.

(2) In addition, what is the magnitude range of events which dominate the seismic hazard, specifically for seismic source zones with upper magnitude cutoffs in the range of 6.5 to 7.0, at accelerations of 0.60g to 1.0g?

The staff also needs additional information on your statement that C_D, the correction factor on ductility, is considered to be frequency independent.

(3) Give evidence to support your statement that C_D is considered to be frequency independent. Provide the specific C_D factors associated with the 8.54 HZ model structure frequency for Tables 4-4 and 4-5 included in the Structural Mechanics Associates seismic fragility analysis.

(4) If the values of C_D from question (3) are lower than the C_D you have used (C_D = 1.3), provide justification for your assumptions.